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NIGERIAN RESEARCH REACTOR-1 CORE NEUTRONICS CALCULATIONS

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ABSTRACT
Feasibility studies for the conversion of the Nigerian Research Reactor-1 (NIRR-1) have been performed using WIMS and CITATION codes at the Center for Energy Research and Training (CERT), Ahmadu Bello University, Zaria Kaduna State. In this work, the core neutronics calculation of NIRR-1 concerning mass loading of U-235 in the core, shut down margin (SDM), safety reactivity factor (SRF), control rod worth, and control rod critical depth of insertion were investigated at low enrichment. Two fuel types (UAl_4 and UO_2) were considered and the uranium densities required for the conversion of NIRR-1 core to low enrichment were computed to be 1201g/cc with 20% enrichment, 1144g/cc with 19.75% enrichment, 1274g/cc with 15% enrichment, 1448g/cc with 10% enrichment for UAl_4 fuel type and 1141g/cc with 20% enrichment, 1144g/cc with 19.75% enrichment, 1216g/cc with 15% enrichment, and 1389g/cc with 10% enrichment for UO_2 fuel type. Significantly, higher uranium densities are required to convert NIRR-1 from HEU to LEU – indicating a drastic review of the NIRR-1 core.

KEYWORDS: Neutronics, CITATION, Reactivity, Reactor, MNSR, WIMS

INTRODUCTION

As part of the on-going global effort to convert HEU reactor cores to Low Enriched Uranium (LEU) cores under the Reduced Enrichment for Research and Test Reactors (RERTR) program, there is need to study the possibilities of converting NIRR-1 core to less than 20% enrichment. Works have been done on Research and Test Reactor core conversion around the globe. In some proposed models, the total number of fuel pins (Khamis and Khattab, 1999) and the core radius/height ratio (Matos and Lell, 2005) has been drastically changed. This brings about noticeable changes in the relative flux values for both inner and outer irradiation sites. In this work, HEU and LEU cores are analyzed using the present UAl_4 fuel and a potential LEU fuel (UO_2 clad in zircalloy) are considered. The existing HEU core was also analyzed to validate the reactor model used. A significant feature of this work is the preservation of the technical and the geometric specification of the reactor so as to maintain the original designed thermal hydraulic of the reactor. A detailed description of the HEU core of NIRR-1 can be found in the Final Safety Analysis Report (FSAR); (Azande, *et al*, 2010; SAR, 2005).

THEORITICAL CONSIDERATION

The resonance treatment in WIMS is based on equivalence theorems, which relate a library of resonance integrals for each resonance absorber in each group to a particular heterogeneous problem. The equivalence theorems employed take account of non-homogeneous moderators which might be mixed with the fuel such as oxygen or carbon; (Askew *et al*, 1966). A first order correction for the interaction associated with the presence of several resonant nuclides was also included.

The library of homogeneous resonance integrals has been formed through the use of the Spin Dependence Recombination (SDR) code; (Brissenden and Durston, 1965). The SDR code solves by numerical methods the slowing down problem associated with moderators homogeneously mixed with resonant nuclides. The code solves the Chernick and Vernon integral equation (Rothenstein, 1960) in the form

$$V_o F_o(E) = P_{oo}(E)\int_E^{E/\alpha_o} \frac{\Sigma_{so} V_o F_o(E')}{\Sigma_o E'(1-\alpha_o)} + dE' + \sum_{i\neq 0} P_{io}(E)\int_E^{E/\alpha_o} \frac{\Sigma_{si} V_i F_i(E')}{\Sigma_i E'(1-\alpha_i)} dE' \qquad (1)$$

•

$P_{oo}(E)$ is the self collision probability in fuel region, $F_o(E')$ is the collision density, $F_i(E)$ is collision density for a general moderator region *i*, and P_{io} is the probability of a neutron scattered in region *i* suffering its next collision in region *o*.

In order to derive the equivalence principle, it is usual to assume that the resonances are narrow and well separated and the fluxes $\phi_i(E')$ for energies outside the resonances may be represented by

$$\phi_i(E') = \frac{d_i}{E'} \qquad (2)$$

In which d_i is a disadvantage factor, which is assumed invariant with energy such that

$$d_i = \phi_i(E')/\phi_o(E') \qquad (3)$$

If it is further assumed that the probabilities $P_{io}(E)$ can be represented by the type of approximation attributed to (Bonalumi, 1961) in the form

$$P_{io} = \frac{V_o\Sigma_o}{V_i\Sigma_i}P_{oi} = (1-P_{oo})G_{oi}\frac{V_o\Sigma_o}{V_i\Sigma_o} \qquad (4)$$

G_{oi} is the probability of those neutrons escaping collision in region *o* suffer their next collision in region *i*. Applying equations (2), (3), and (4) in equation (1) with the narrow resonance approximation gives

$$\phi_o(E) = \left[P_{oo}\frac{\Sigma_{po}}{\Sigma_o} + (1-P_{oo})\sum_i d_i G_{oi}\right]\frac{1}{E} \qquad (5)$$

In the case where d_i is unity and $\sum_i G_{oi} = I$, gives

$$I = \int \left[P_{oo}\frac{\Sigma_{po}}{\Sigma_o} + (1-P_{oo})\right]\sigma_{oo}\frac{dE}{E} \qquad (6)$$

Σ_{po} is potential scattering cross-section of the fuel with admixed moderator and σ_{oo} is the resonant macroscopic absorption cross-section.
For isolated bodies, the Bell modification (Bell, 1959) of the Wigner relational approximation gives

$$I = \int \frac{\Sigma_{po} + a\Sigma_e}{\Sigma_o + a\Sigma_e} + (1-P_{oo})\,\sigma_{oo}\frac{dE}{E} \qquad (7)$$

α is the bell factor. The methods adopted or extending the equivalence theorem for isolated rods to infinite arrays were suggested by Leslie and Jonsson, 1964.

METHODOLOGY

A simplified scheme of WIMS and CITATION codes (Azande *et al*, 2010; Balogun, 2003) were employed to generate group constants for the fuel region and non fuel regions such as the control rod, control rod follower region as well as reflectors. Unit cell calculations based on Wigner-Seitz cell modeling was used. The core lattice was represented in the form of a super cell; (Iqbal *et al*, (2002). A two-dimensional CITATION input (that is r-z cylindrical geometry) (Balogun, 2003) was prepared with group constants associated with homogenized regions, including the control rod. Although the group constants associated with the control rod was incorporated, the base input data was prepared such that water completely occupies the channel in which the rod travels. Data were introduced on four levels including

1. Reactor level: data used to describe general calculation options such as
   - the number of components forming the reactor,
   - the number of energy groups used in the calculations,
   - the total number of materials of which the reactor components are formed,
   - some factors indicating to the system how to model the reactor angularly.

2. Component level: data used to describe the component itself such as
   - the component number;
   - the number of pieces forming the component;
   - for each piece forming the component data include the following parameters:
     - the internal radius of the piece (mm),
     - the external radius of the piece (mm),
     - the height of the piece (mm).
     - some other parameters that can represent holes and other forms of the piece other than the cylindrical one.

3. Material level: data used to describe the elements forming the material. This level is mainly directed to WIMSD4 to enable the generation of the group constants for the different materials.

4. CITATION level: the data here are free format numbers representing the user options to be applied in CITATION code during core calculations. This level includes also control cards options of the code CITATION. CITATION input file would automatically be compiled based on the indications of the data presented here.

Once the reactor pieces, materials, and CITATION options are specified, an automatic processing of these data is performed, and a modeled reactor is generated. When the modeled reactor is generated, material data and CITATION options are used to generate cross sections and group constants by the automatic running of WIMSD4. Once WIMS is run, an automatic compilation of the CITATION input file is performed; hence CITATION runs automatically to generate an output file with the options specified in the CITATION file.

The initial excess reactivity ρ_{ex} is computed automatically as

$$\rho_{ex} = \frac{k_{eff} - 1}{k_{eff}} \qquad (8)$$

and the result is displayed on the screen. k_{eff} is the effective multiplication factor of the reactor. Flux and power distributions are also automatically generated. The generation and transfer of the group constants form WIMSD4 input file in WIMS format to the input file of CITATION and in CITATION format is achieved automatically. The model used in WIMSD4 to generate the Group Constant is automatically selected too.

RESULTS AND DISCUSSION

The results from Tables 1 and 2 show SDM values within the range of 2.0 and 3.5 mk which is in good agreement with design specification for shutdown margin of MNSR. The excess reactivity values obtained are also in good agreement with the design specifications of (3.5 - 4.0) mk; (Chengzhen, 1993).

Table 1: Results for UAl_4 fuel at low enrichments

Enrichment (%)	^{235}U core loading (g)	S.R.F	Rod worth (mk)	S.D.M (mk)	Excess Reactivity (mk)	Critical Depth (cm)
05.00	2505	1.24	4.36	0.84	3.52	18.69
10.00	1448	1.59	5.63	2.08	3.56	13.66
15.00	1274	1.70	5.99	2.46	3.53	12.94
20.00	1201	1.60	6.16	2.18	3.86	13.66

Table 2: Results for UO_2 fuel at low enrichments

Enrichment (%)	^{235}U core loading (g)	S.R.F	Rod Worth (mk)	S.D.M (mk)	Excess Reactivity (mk)	Critical Depth (mm)
05.00	2505	1.24	4.36	0.84	3.52	18.69
10.00	1389	1.67	5.87	2.35	3.52	12.94
15.00	1216	1.77	6.27	2.73	3.55	12.22
20.00	1141	1.83	6.48	2.93	3.55	12.22

When the value of multiplication factor k_{eff} is greater than unity, the reactor assembly is said to be supercritical and when less than unity, it is a subcritical assembly. For fission chain reaction to be just continuing at a steady state, k_{eff} has to be unity. Expectations are that when the control rod is fully withdrawn, the super-criticality should be attained while sub-criticality should be attained when the control rod is fully inserted. It is therefore, desirable that criticality should be attained about midway between full insertion and full withdrawal of the control rod so as to allow for easy control and regulation of reactivity. The results obtained show critical depths which approximately satisfied this expectation. The safety reactivity factor (SRF) of greater than 1.5 was obtained which is in agreement with Balogun, (2003).

CONCLUSION

In this core neutronics calculation of NIRR-1 at LEU core using WIMS and CITATION codes, results are in good agreement with HEU results and design specifications.

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